Nuclear data in the problem
of fission reactor decommissioning.

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In this report a review is given of the works published in Russia during last several years and devoted to the problem of nuclear data and calculations of nuclear facility activation for fission reactor decommissioning.

1.General concept of reactor decommissioning in the framework of the former Council of Mutual Economic Assistance was published in the work [1] in 1990. According to this work during period up to 2000 year it was planned to decommission 10-15 reactor units of nuclear power stations. The preliminary estimates shows that for complete removal of nuclear power reactor WWER-1000, including the dismantling of inventory and systems and the destroying of buildings,the removal of radioactive wastes from reactor site, the expeanses amounts to 30% of the total cost of NPS. These expeanses depend strongly on the amount of radioactive wastes, which are within $(5-20)\cdot10^3$ tons,the methods its treatment and separation from materials, which can be used in allowable limits.

That is why the main objectives in the NPS decommissioning is not only minimization of collective irradiation dose and also minimization of radioactive waste amounts.

In this connection it is especially important to have a reliable calculation of the reactor inventory activation and radiation fields in the places of the dismantling and wrapping of inventory. A number of works are devoted to these problems [2-5]. The main results of the calculations from these works are given below.

2.In the work [2] the calculational estimates are made of the space and time dependencies of specific activity and rate of equivalent dose for model, which imitates reactor vessel wall of WWER type reactor (layer of steel is 19 cm thick). For the activation calculations the activation data library formed by the authors was used. The library includes group activation cross sections and decay characteristics of 46 radionuclides. The data of the library have 40-group presentation as in CASK-system.

The analysis of the results of calculations allows to do the following conclusions:

- main contribution to the total activity of WWER vessel and dose rate of activation product gamma-radiation. in 2 years after shutdown give radionuclides, arising from the following reactions: $^{59}Co(n,\gamma)^{60}Co$, $^{54}Fe(n,p)^{54}Mn$.

- space distribution of the activity of radionuclides in reactor vessel wall is determined by space-energy distribution of neutrons in it during reactor operation.

So the activity of ^{60}Co, arising from thermal neutron radiative capture, decreases rapidly in vessel wall, and the space distribution of ^{54}Mn, arising from threshold reaction, correlates with space distribution of fast neutron flux in iron, and exponentionaly decreases.

The work [3] is devoted to prognosis of neutron induced activity and field of gamma-radiation of ^{60}Co in the shield of IRT reactor of MIEP for decommissioning. The code ANISN with DLC-23/CASK library was used.

The conclusion is given in this work that induced activity and radiation field depend on the time power distribution during reactor operation,reactor construction and shield,element composition of structural and shielding materials, including tracer elements so as cobalt,europium, which very often determine induced activity. It is noted that content of tracer elements varies usually within large range. That is why for every reactor in process of preparation for decommissioning it is necessary to calculate induced activity and gamma-radiation of activation products.

3.The work [4] contains the result of careful investigation of induced activity in concrete used for reactor shield. It is noted that the calculation of concrete activity is not easy task because of lack of information about concentration of elements giving long-lived radionuclides ,in particular cobalt, europium, cesium and tantalum. The work contains the results of concentration measurements of these elements for more often used concrete component: limestone, granite, serpentite, and their activity induced during reactor operation. It is shown that in 250 days after shut-down the activity of these materials are determined by long-lived radionuclides: in limestone and granite - ^{152}Eu, ^{60}Co, ^{134}Cs, ^{45}Sc and ^{182}Ta, in serpentite - ^{60}Co.

For comparison the calculations were made of time behavior of specific induced activity for the materials mentioned above in 30 years of reactor operation .

These calculations show that the average activity of limestone ten times lower in comparison with granite activity and 100 times lower than serpentite activity and equal to 1.66×10^{-6} in 0.5 years after shutdown.

The main radionuclides which determine activity of limestones are ^{152}Eu, ^{154}Eu, ^{60}Co; from them the contribution of ^{152}Eu is the greatest and changes from 60% in 0.5 years to 75% in 20 years after shutdown.The average activity of granite in 0.5 years after shutdown is $4.55\times10^{-5} Bk\times g^{-1}/m^{-2}\times s^{-1}$ and in 20 years decreases only 3 times.

High concentration of cobalt in serpentite (1×10^{-2} mas %) determines its induced activity. The induced activity of cement used for concrete was measured as well. The calculation allows to make conclusion that the activity of concrete with limestone is determined by cement activity. The activity of concrete with granite and serpentite are determined by these materials. It is noted that there are large uncertainties in concentration of elements which determine long-lived induced activity of rocks.

4. For development and improvement of codes and constants it is very important to do calculational investigations of test models of reactor shield of various types. In this connection one should note the work by Savitsky [6] on development of fast neutron reactor shield model. Such investigations give possibility for those who develops code systems and data libraries to check reliability of codes and constants. The models are created having in mind the shield composition of reactors in operation and the result of the calculation of radiation fields for this models can be related to some extent with real shield construction.

5. In the connection with nuclear data problem a considerable attention is paid to the analysis of sensitivity to changes in input parameters. Several codes are developed and used for this purpose, for example SWANLAKE, ZAKAT and its modifications. The modified version of SWANLAKE, which is called SWANLAKE-ACT [3] is based on linear perturbation theory for sensitivity coefficient calculation of the functionals of activation product γ-radiation field to neutron reaction cross sections, nuclear concentration of activatied isotopes and decay constants of nuclei-activation products.

Using the code SWANLAKE-ACT in the work mentioned the calculations were made of energy dependence of sensitivity to microscopic activation cross sections,and rate of equivalent dose.

6. For further development of constant systems it is very important to work out the principles of creating the purpose-oriented libraries.

The specialized nuclear constant library for the calculation of fast neutron power reactor (PROBA-BN)[6] is formed on the base of multigroup library VITANIN-C.

The specialized group libraries used in calculations give additional uncertainties, connected with transition from multigroup presentation of cross sections to group presentation (uncertainty of cross section averaging).

If to accept that uncertainty of cross section averaging should not exceed uncertainty of constant component of radiation field calculation, then the decreasing of uncertainty existing as a result of averaging cross sections may be achieved by using space dependent group constants which contain different group constants for the same isotopes but in different physical zone of nuclear facilities.At the same time the decreasing of uncertainty of cross section averaging may be achieved by optimization of group structure of library by means of using additional information on sensitivity of functionals of radiation field to interaction cross sections.

7. In conclusion one should note that in Russia for engineering calculation of nuclear reactors and shield the codes ROZ (one-demensional calculation) and RADUGA (two-dimensional calculation) and similar foreign codes ANISN and DOT are widely used. These codes are connected with the group constant system BNAB. At the present time in addition to the version BNAB-78 the new version BNAB-90 was implemented into practical calculations. The BNAB-90 includes multigroup presentation with maximum number 300. The multigroup presentation is used in the cases when it is important from point of view of calculation accuracy. The aim is in process of nuclear data averaging not to lose the accuracy achieved in pointwise microscopic cross sections. The data of BNAB system can be prepared in ANISN format and the system was adapted on PC/AT. The BNAB-90 system is used for the calculations of shield and activation of reactor inventory. For this purpose the library of activation data is formed, which includes cross sections for more 3000 reactions, decay data and other data. It is planned the further evaluation and testing of activation cross sections of the library.

List of works under review.

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[2] Mashkovich V.P. New investigations in the field of radiation shield physics.- Atomnaya Energia, 1990, v.68, issue 6, pp 427-431.

[3] Dovbenko A.A., Kudrjavtseva A.V., Stroganov A.A. Nuclear data for the calculation of photon radiation field of activation products in the task of thermal neutron NPS decommissioning. - VANT, Ser.Yadernye Konstanty, 1990, issue 1, pp 106-119.

[4] Dovbenko A.A., Kudrjavtseva A.V., Mashkovich V.P., Neretin V.A., Petrov V.N., Stroganov A.A. Induced activity and gamma-radiation field of activation products in the shield of IRT reactor MIEP for decommissioning. - Atomnaya Energia, 1991, v.71, issue 5, pp 431-436.

[5] Lavdansky P.A., Nazarov V.M., Stefanov N.I., Frontas'eva M.V. Induced activity of concrete used for the shielding of nuclear facilities. - Atomnaya Energia, 1988, v.64, issue 6, pp 419-442.

[6] Boljatko V.V., Mashkovich V.P., Neretin V.A., Savitsky V.I., Stroganov A.A. Formation of the purpose-oriented constant system on the example of fast reactor shield.- Atomnaya Energia, 1989, v.66, issue 3, pp 199-203.